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Subject: Supplemental Information regarding License Amendment Application to Revise Technical Specification 3/4.4.5, "Reactor Coolant System - Steam Generators," to Permit One-Time Extension of Steam Generator Tube Inservice Inspection Interval (TAC No. MC1573)

Ladies and Gentlemen:

By letter dated December 16, 2003 (Serial Number 3000), the FirstEnergy Nuclear Operating Company (FENOC) submitted an application for amendment of the Operating License, Appendix A, Technical Specifications (TS) for the Davis-Besse Nuclear Power Station (DBNPS). The proposed amendment would revise Technical Specification (TS) 3/4.4.5, "Reactor Coolant System - Steam Generators," to allow a one-time extension of the steam generator tube inservice inspection interval. By letter dated January 23, 2004 (Log Number 6150), the NRC requested additional information regarding the proposed amendment. FENOC responded to this request by letter dated January 29, 2004 (Serial Number 3023). A follow-up request for additional information was sent by the NRC via e-mail on February 4, 2004. The response to this request was discussed during a telephone call between the FENOC and NRC staffs on February 9, 2004. A summary of the information provided by FENOC during this discussion is documented in Attachment 1.

Should you have any questions or require additional information, please contact Mr. Gregory A. Dunn, Manager - Regulatory Affairs, at (419) 321-8450.

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The statements contained in this submittal, including its associated attachments, are true and correct to the best of my knowledge and belief. I declare under penalty of perjury that the foregoing is true and correct.

Executed on: 2/13/04.

By:

Mark B. Bezilla, Vice President - Nuclear

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Attachments

cc: Regional Administrator, NRC Region III
J. B. Hopkins, DB-1 NRC/NRR Senior Project Manager
C. S. Thomas, DB-1 NRC Senior Resident Inspector
Utility Radiological Safety Board

SUPPLEMENTAL INFORMATION REGARDING LICENSE AMENDMENT APPLICATION TO REVISE TECHNICAL SPECIFICATION 3/4.4.5, "REACTOR COOLANT SYSTEM - STEAM GENERATORS," TO PERMIT ONE-TIME EXTENSION OF STEAM GENERATOR TUBE INSERVICE INSPECTION INTERVAL

During inservice inspection of the Davis-Besse Nuclear Power Station (DBNPS) once-through steam generators (OTSGs) during the thirteenth refueling outage (13RFO), an axial crack was identified in OTSG tube 2A-Row 63 Tube 78. The axial crack in OTSG tube 2A-Row 63 Tube 78 was in the roll expanded region of the tube, in the upper roll, but not in the roll transition region itself. This particular crack is not characteristic (length, depth, voltage) of what has historically been observed in once-through steam generators at the roll transition area. Roll transition cracking in the OTSGs is typically <1 volt Plus-point, confined within the transition (<0.25"), and with average depths much less than 90% through-wall (TW). Because of its location within the roll joint and above the transition, this indication had no impact on the DBNPS structural or leakage integrity at end-of-cycle (EOC) 13. Figure 1 provides the Plus-point C-scan plot of this indication/region. The thirteenth refueling outage (13RFO) inspection was the first inspection of this tube in this region after 15.8 effective full power years (EFPY) of operation at DBNPS with an eddy current technique that is capable of reliably detecting indications of this degradation type/location. The fact that the profile of the indication is >90%TW for almost its entire length is not totally unexpected (Figure 2). Upper 95th percentile growth rates for these indications in OTSGs is approximately 13%TW/EFPY and 0.2"/EFPY, indicating approximately 4-5 cycles of growth for this particular indication. The bobbin coil data for this tube was also reviewed from both the 13RFO and 12RFO inspections and indicated the presence of a signal that can be matched to that of the Single Axial Indication (SAI) reported in 13RFO from the Plus-point inspection (Figures 3 and 4). This data also indicates that the growth did not occur over a single cycle and is consistent with the historical growth rates of this degradation.

The history of upper roll/tube end region inspections and primary water stress corrosion cracking (PWSCC) at the DBNPS is summarized below in Table 1. After 15.8 EFPY of operation at the DBNPS only one incidence of classical roll transition axial PWSCC has been detected. During 10RFO, tube 2-A R58T119 was reported as having an SAI in the roll transition region. This tube was

determined to have been re-rolled in the factory, after the full vessel heat treatment, but prior to SG shipment. A 13-inch long section of the tube was removed during the 10RFO outage, which occurred at 10.6 EFPY. The eddy current indication was confirmed to be an axial crack caused by primary water stress corrosion cracking. The crack was 0.092 inches long, had a maximum depth of 78% through wall, and an average depth of 61% through wall. The crack had no circumferential extent or branching. Currently, there are six non-stress relieved rolled joints that remain inservice in the DBNPS OTSGs. Because these re-rolls were performed after heat treatment, the stress state in these tubes is believed to be considerably higher and thus more susceptible to roll transition PWSCC compared to the stress relieved roll transitions. Since these locations are likely to crack first, they have been inspected with Plus-point in the roll region every outage since 10RFO. No other indications of roll transition cracking have been found at the DBNPS since discovery of the roll transition crack discussed above. This observation is consistent with the fact that the DBNPS original rolls are both stress relieved and thermally treated by the stress relief annealing of the entire steam generator.

Table 1
DBNPS Plus-Point Probe Inspection History
Upper Roll Region in Both OTSGs

Outage Date	Outage ID	Approximate Cumulative EFPY	UTE / RT Plus-Point Sample	Indications Detected Above the Upper Tube End Roll Transition each outage	Upper Roll Transition Axial Cracks
April 1996	10RFO	10.6	<1%	0	1*
April 1998	11RFO	12.3	~20%**	5	0
April 2000	12RFO	14.1	~21%**	12	0
Feb 2002	13RFO	15.8	~57%**	47	0

* Non-stress relieved roll joint.

** These percentages represent the ratio of the number of inspected upper tube ends to the total number of unsleeved tubes. The percentages do not total 100% because tubes that have been removed from service by plugging were not inspected. All in-service upper tube ends have been inspected within the last three eddy current examinations.

The last two OTSG operational assessments performed for DBNPS have considered roll transition axial PWSCC as a potential to occur during the upcoming cycle. In both outages (12RFO and 13RFO), no indications were actually detected, indicating that the analysis method was conservative. The assessments utilized a multi-cycle Monte Carlo model to project a population of indications, their severity (i.e. depth), and the potential contribution to leakage during a main steam line break (MSLB) event. In the current operational assessment, the number of detected roll transition cracks at EOC 13 was conservatively taken as 2, which provided a conservative initiation function for the multi-cycle model, even though no detected cracks exactly fit into this category. The projected EOC 14 95-95 probability/confidence level steam line break leak rate from a multi-cycle Monte Carlo analyses of roll transition axial PWSCC, where inspection scope is explicitly considered in the model, is 0.0 gpm.

The planned scope for the next inspection (mid-cycle 14) is approximately a 43% sample of the upper roll/tube end region of both OTSGs at DBNPS. This will complete a 100% inspection of the population in less than one cycle of operation (~1.0 EFPY). Using a deterministic worst case approach, a flaw of approximately 95%TW could be postulated to exist in this 43% sample, since about half of the tubes in the sample will be in operation for about 4.5 EFPY at the proposed mid-cycle inspection. This assumes a worst case Plus-point probability of detection (POD) of 0.89 at 37%TW at 90% confidence level and an upper 95th percentile growth rate of 13%TW/EFPY for the half of the sample that was inspected last in 11RFO. The other half of the 43% sample will have only gone about 2.7 cycles of operation since being inspected during 12RFO. However, using a probabilistic Monte Carlo method, the probability of a flaw of this magnitude existing in the 43% sample is very low. As previously stated, this method had conservatively predicted flaws at EOC 12 and 13 while none actually were detected, and projects zero accident leakage at EOC 14, which is 0.8 EFPY beyond the scheduled mid-cycle outage. Based on these evaluations, it is not expected for either a significant number of indications or an indication of large magnitude to be detected at the planned mid-cycle 14 outage. However, if a roll transition flaw is discovered during the mid-cycle inspection, the requirements of NEI 97-06, *Steam Generator Program Guidelines,* and Revision 6 of the EPRI *Pressurized Water Reactor Steam Generator Examination Guidelines* will be followed for determining the expansion of the inspection to the unscheduled tube population.

Figure 1 – C-scan of R63T78 in 13RFO

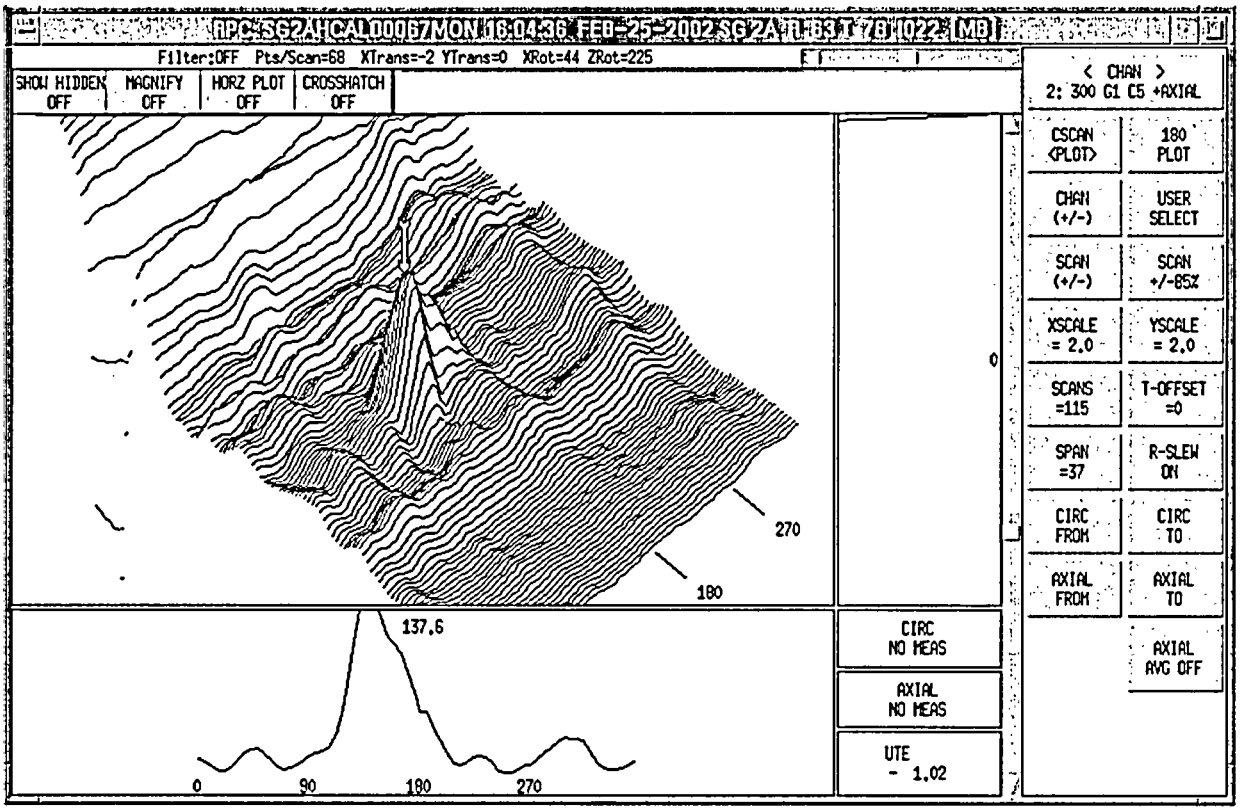


Figure 2 – NDE Profile of R63T78

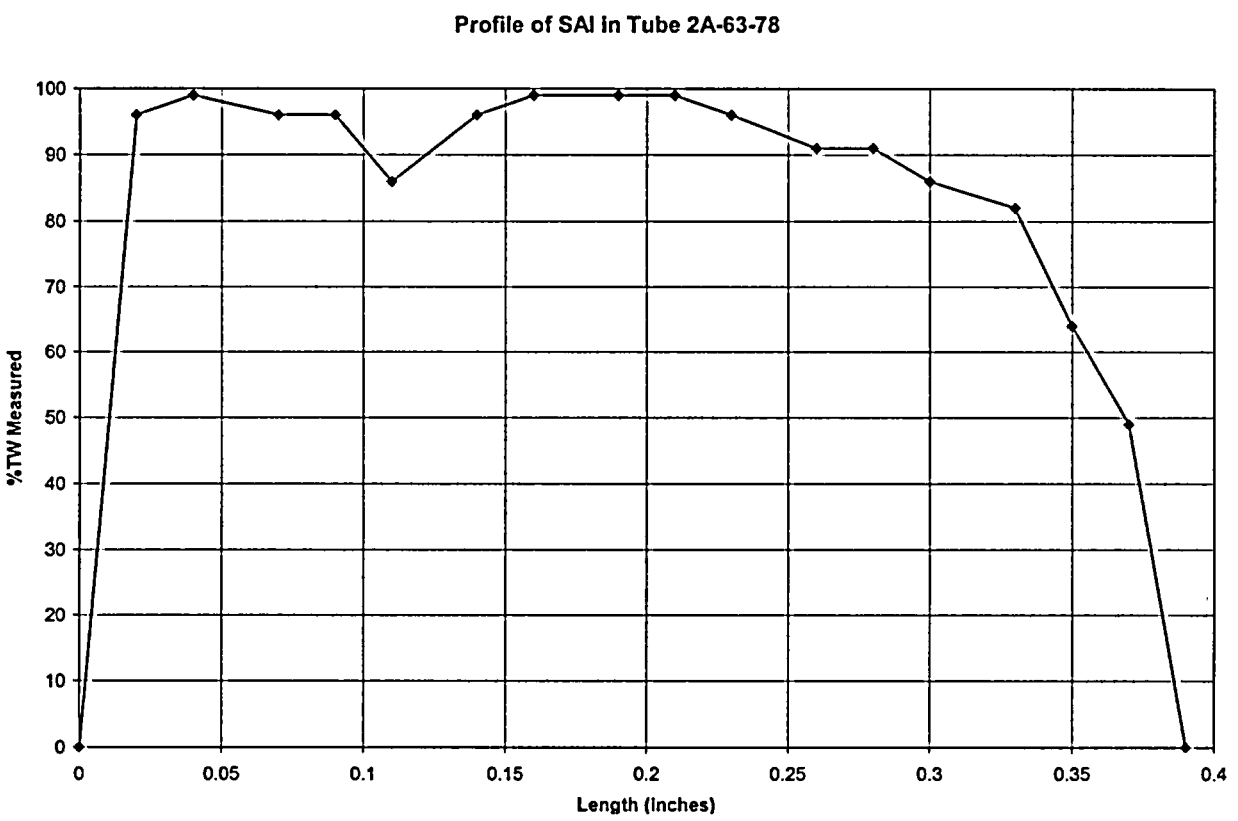


Figure 3 – 12RFO Bobbin Coil Graphic of R63T78 Roll Region

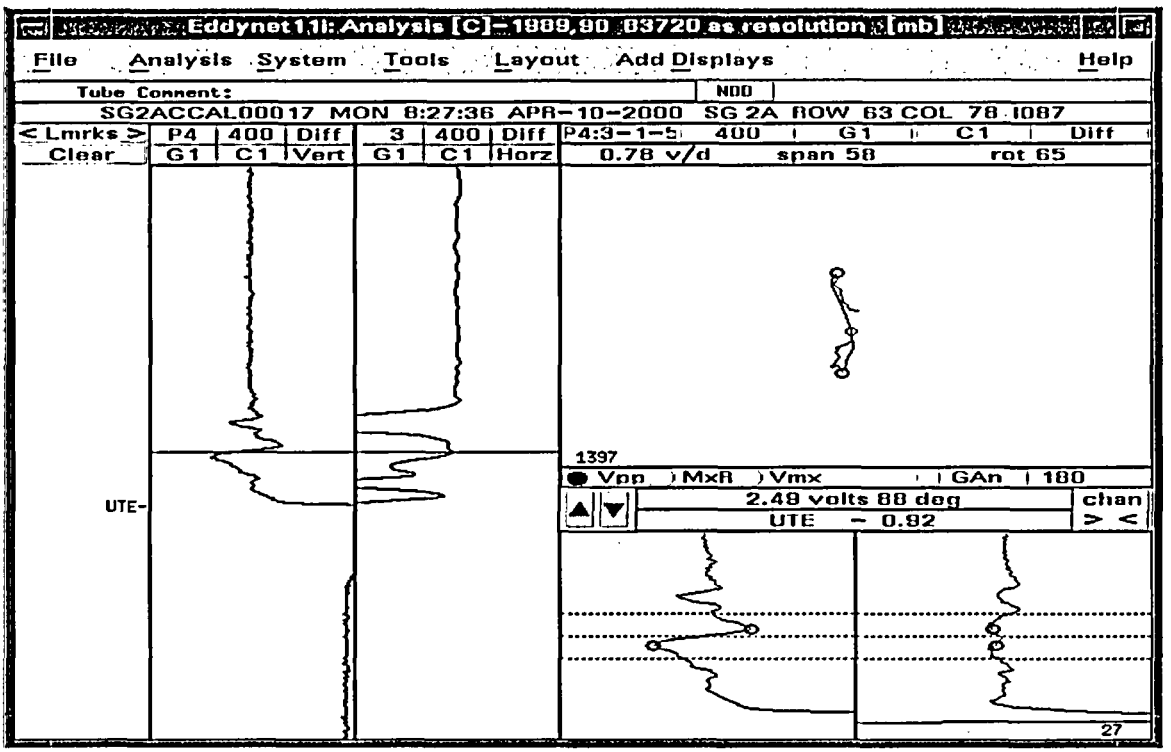
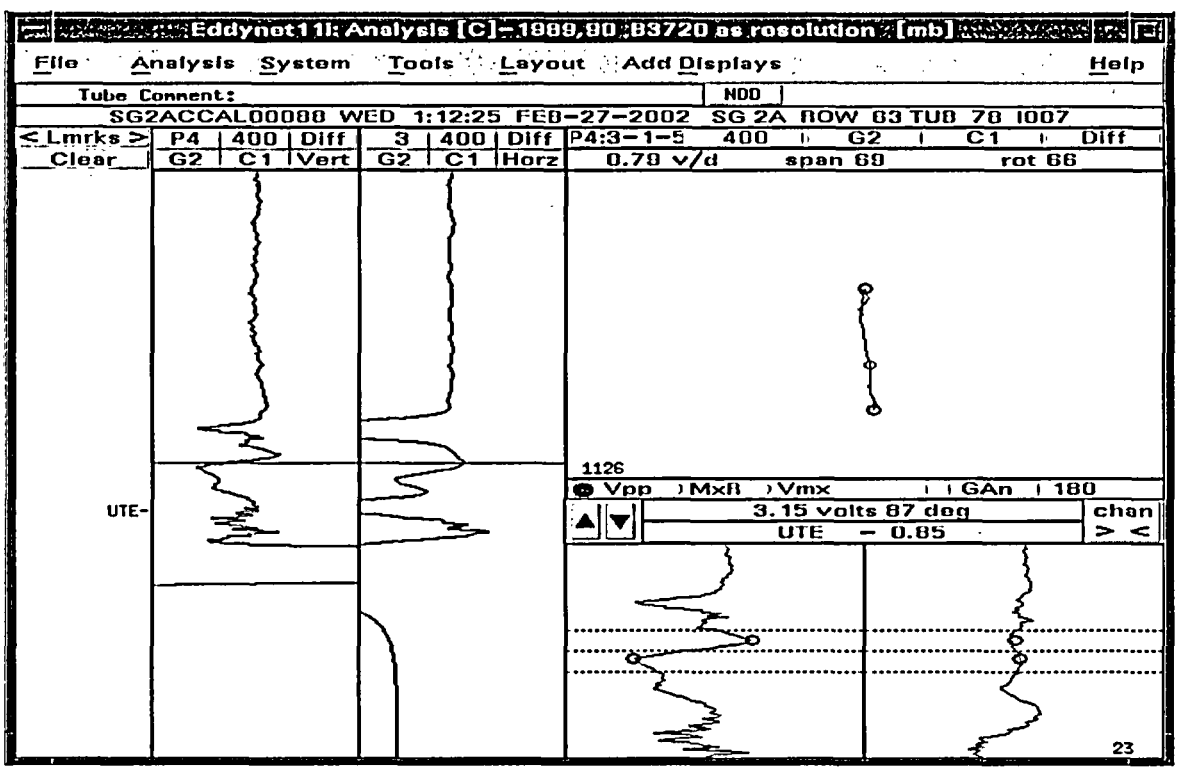


Figure 4 – 13RFO Bobbin Coil Graphic of R63T78 Roll Region



COMMITMENT LIST

The following list identifies those actions committed to by the Davis-Besse Nuclear Power Station (DBNPS) in this document. Any other actions discussed in the submittal represent intended or planned actions by the DBNPS. They are described only for information and are not regulatory commitments. Please notify the Manager – Regulatory Affairs (419-321-8450) at the DBNPS of any questions regarding this document or any associated regulatory commitments.

COMMITMENT	DUE DATE
If a roll transition flaw is discovered during the mid-cycle inspection, the requirements of NEI 97-06, *Steam Generator Program Guidelines,* and Revision 6 of the EPRI *Pressurized Water Reactor Steam Generator Examination Guidelines* will be followed for determining the expansion of the inspection to the unscheduled tube population.	During the Mid-Cycle 14 Outage.